U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Staff Feedback on NuScale's Response to RAI 9666 Related to the EPZ Sizing Methodology Topical Report (TR-0915-17772)

Public Meeting

August 6, 2019

Abbreviations and Acronyms

- **CDF –** core damage frequency
- **CEMP -** comprehensive emergency management plans
- **CNV** – containment vessel
- **CSDRS -** certified seismic design response spectrum
- **DCA** – design certification application
- **DID** – defense-in-depth
- **EPZ** – emergency planning zone
- **HCLPF** - high confidence low probability of failure
- **LRF** – large release frequency
- **LWR** – light water reactor
- **NPM** – NuScale power module
- **NPP** – nuclear power plant
- **PAG –** protective action guide
- **PGA –** peak ground acceleration
- **PRA –** probabilistic risk assessment
- **PSHA** – probabilistic seismic hazards analysis
- **RAI** – request for additional information
- **RPV** – reactor pressure vessel
- **SER** – safety evaluation report
- **SMA** – seismic margins assessment
- **SSC** – structures, systems, and components
- **TR** – topical report

RAI 9666, Question 1.05-34: Acceptability of a DC PRA for Risk Informed Applications

- **RAI Response-** the applicant will need to demonstrate the technical acceptability of the PRA is sufficient to support risk-informed decision making.

- **Staff feedback –** The response is satisfactory and staff will incorporate a condition of use in the SER on acceptability, including the impact of uncertainties on numerical thresholds.

RAI 9666, Question 1.05-36: a) How insights from Level 2 PRA are considered, and b) statement that severe accident phenomena are not credible

- **RAI response**
  - Insights will be considered for accident sequences that screen in.
  - TR updates:
    - Removes references to severe accident phenomena not being credible
    - Adds, “methodology does not utilize the DCA PRA and assessment of containment integrity will be performed with the PRA which is associated with the application.” (Section 3.4.3)
    - Adds, “application of the EPZ methodology should consider the assessment of severe accident phenomena available at the time.” (Section 3.8.2)
- **Staff feedback**
  - Level 2 PRA insights issue will be addressed with RAI Question 1.05-38
  - Clarification requested regarding added text relative to finality of design certifications

RAI 9666, Question 1.05-37: Evaluation of module drop scenarios

- In TR, accidents are "less severe" if containment does not fail.
  - “Less severe” accidents are evaluated against the early phase PAGs.
- In TR, accidents are “more severe” if containment fails or if the containment is bypassed.
  - “More Severe” accidents are compared against the 200 rem LRF criterion.
- Approach implies less severe accidents are more frequent/more severe accidents are less frequent.
- Appropriate for current LWR designs evaluated in NUREG 0396.
- Module drops are the most likely cause of core damage.
- Drop of a fully assembled module is assumed to cause a containment breach and is evaluated against the 200 rem criterion.
- Drop of the upper CNV and upper RPV on fuel located in the RFT is screened.
- The staff expected module drops to be evaluated to the early phase PAGs.

RAI 9666, Question 1.05-37: continued

- **RAI response:**
  - Drop of upper CNV and upper RPV screened from the PRA (no Large release) and therefore, screened from the methodology.
- **Staff response:**
  - Drop of an intact module and drop of the upper RPV and CNV on fuel are more likely scenarios but evaluated against the more severe/less likely criterion.
  - NUREG 0396 (page I-9) states design basis accidents and less severe core-melt accidents should be considered for protective actions.
  - To provide the same level of protection as NUREG 0396, module drops of fully assembled and partially assembled module should be assessed against the PAGs since they are most likely.

RAI 9666, Question 1.05-38: Consistency of proposed criteria for “low Defense-in-Depth” with Commission expectations for advanced light water reactors

- **RAI Response**
  - DID analysis is aimed at establishing PRA technical adequacy
  - If only an approved, technically adequate PRA can be used to support EPZ method, purpose of the DID analysis becomes extraneous and unnecessary
  - Based on response to Question 01.05-34, DID evaluation section of TR should be deleted
- **Staff Feedback**
  - Response appears to be inconsistent with NRC’s PRA policy statement
  - Confidence that “more severe” accidents (i.e. containment bypass sequences) would not produce significant off-site consequences would provide DID

References on Defense-in-depth

Regulatory Guide 1.174, Revision 3

The defense-in-depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena that, because they are unknown or unforeseen, are not reflected in either the PRA or traditional engineering analyses.

1995 Commission Policy Statement on Use of PRA Methods in Nuclear Regulatory Activities

The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

NUREG/KM-0009, "Historical Review and Observations of Defense-in-Depth"

"... the ultimate purpose of defense-in-depth is to compensate for uncertainty (e.g., uncertainty due to lack of operational experience with new technologies and new design features, uncertainty in the type and magnitude of challenges to safety)."

Defense-in-depth, in the NUREG, is defined as ". . . an element of NRC's safety philosophy that is used to address uncertainty by employing successive measure including safety margins to prevent and mitigate damage if a malfunction, accident or naturally caused event occurs at a nuclear facility."

RAI 9666, Question 1.05-35: Adequacy of the PRA based SMA to determine EPZ size

RAI response:

- As stated in NUREG-1738 (p. 3-37), for high PGA earthquake, it was reasoned that there would be no effective evacuation and many structures would be uninhabitable.

- Pre-planned response as defined in emergency plans would be significantly less effective than an integrated national response, which assesses and accounts for damaged infrastructure over a wide area.

- Seismic events, seismic PRA, and SMA were not considered in determining the 10-mile plume exposure distance for EPZ.

RAI 9666, Question 1.05-35: Adequacy of the PRA based SMA to determine EPZ size (continued)

Staff response:

- Evacuation is site specific.
- The EP rule does not rely on comprehensive emergency management plans (CEMPs). These CEMPs are not reviewed and approved by FEMA.
- NUREG 0396 Appendix III states, no specific design basis accident or Class 9 accident scenario can be isolated as the one for which to plan because each accident would have different consequences. It appears that no accidents or accident initiators were explicitly eliminated on an a-priori basis.

Seismic Risk

- Does the PRA based seismic margins approach (SMA) appropriately consider credible seismic induced core damage events?
- New application for the SMA.
- Difficult to compare SMA results against CDF screening thresholds.

Staff's confirmatory calculations

- Use fragility information provided for NuScale structural SSCs to estimate failure frequency due to seismic loading
- Use hazard curves from Near-Term Task Force Recommendation 2.1 (R2.1) Seismic Hazard reevaluations to model seismic loading of structural SSCs
  - R2.1 hazard results incorporate latest seismic source and ground motion models
  - Detailed geologic siting information for each site allows determination of local site response
- Evaluate distribution of estimated SSC failure frequencies

Table 19.1-35: Structural Fragility Parameters and Results

Structures	A_m (g)	β_r	β_u	HCLPF (g)	Controlling Failure Mode	Assumed consequence
Reactor Building Crane	2.64	0.28	0.39	0.88	Bridge seismic restraint weldment yielding	Core damage / Large Release
Reactor Building Exterior Walls	1.92	0.12	0.33	0.92	Out-of-plane shear	Core damage / Large Release
NPM Supports	1.98	0.12	0.35	0.92	Shear failure of multiple shear lugs	Core damage/Large Release
Bio Shield - horizontal shear flexure -normal operation	11.62	0.28	0.37	3.99	Horizontal shield slab bending failure	Core damage / Large Release
Bio shield - pool wall bolt failure - normal operation	5.37	0.28	0.35	1.91	Shear Failure of pool wall Anchor Bolts	Core damage / Large Release
Bio shield - horizontal shear flexure - double stacked for refueling of adj. model	4.05	0.28	0.41	1.30	Bending failure of both stacked shield slabs	Core damage / Large Release when configuration present
Bio shield - pool wall bolt failure - double stacked for refueling of adj. model	3.05	0.28	0.35	1.08	Shear Failure of pool wall Anchor Bolts	Core damage / Large Release when configuration present
Pool Walls	2.31	0.21	0.33	0.95	Out-of-plane shear	Core damage / Large Release
Crane Support Walls	2.61	0.12	0.34	1.23	Out-of-plane shear	Core damage / Large Release
Bay Walls	2.65	0.12	0.31	1.31	In-plane flexure	Core damage / Large Release
Roof	2.22	0.12	0.26	1.20	In-plane shear	Core damage / Large Release
Basemat	3.57	0.27	0.31	1.38	Out-of-plane shear	Core damage / Large Release

A_m = median seismic capacity; βu = uncertainty in the median seismic capacity; βr = randomness of the fragility evaluation; HCLPF = High-Confidence (95%) of a Low Probability (5%) of Failure, Reference 19.1-57

Crane Fragility

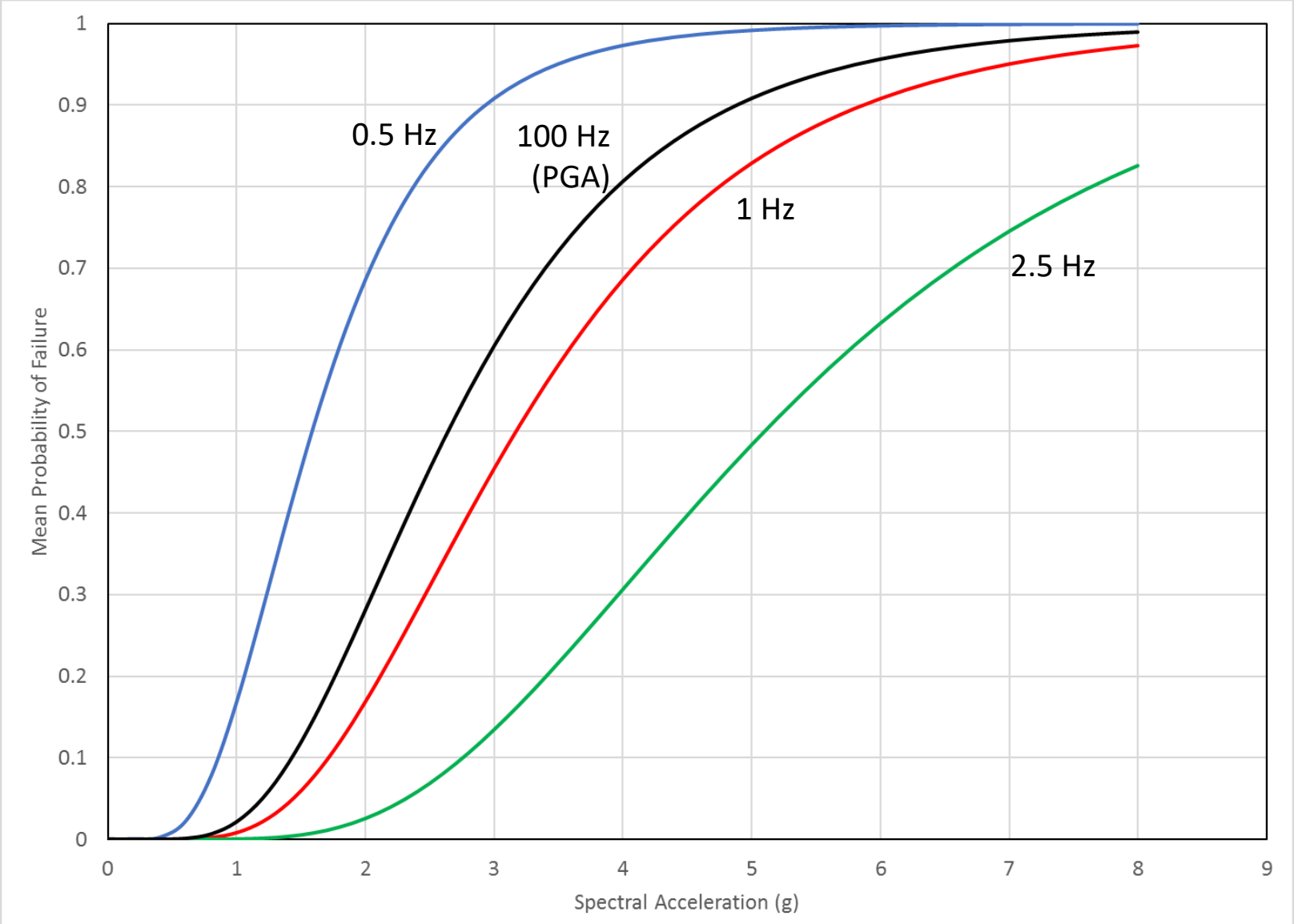
- NuScale Crane HCLPF "High Confidence Low Probability of Failure"
  - HCLPF=0.88 g
  - Uncertainties
    - β_u=0.39
    - β_r=0.28
- HCLPF value is for PGA or 100 Hz ground motions
- The natural or resonance frequency for large structural SSCs ranges from about 0.5 to 2.5 Hz
- Convert PGA HCLPF to HCLPF for 0.5, 1, and 2.5 Hz

Crane Fragility

- Use NuScale Certified Seismic Design Response Spectrum (CSDRS) to develop HCLPF frequency adjustment factors

Spectral Freq (Hz)	Spectral Acc (g)	Freq Adj Factor	HCLPF (g)
0.5	0.3	0.3/0.5=0.6	0.528
1.0	0.6	0.6/0.5=1.2	1.056
2.5	1.0	1.0/0.5=2.0	1.760
100 (PGA)	0.5	0.5/0.5=1	0.880

Crane Fragility for Various Spectral Frequencies



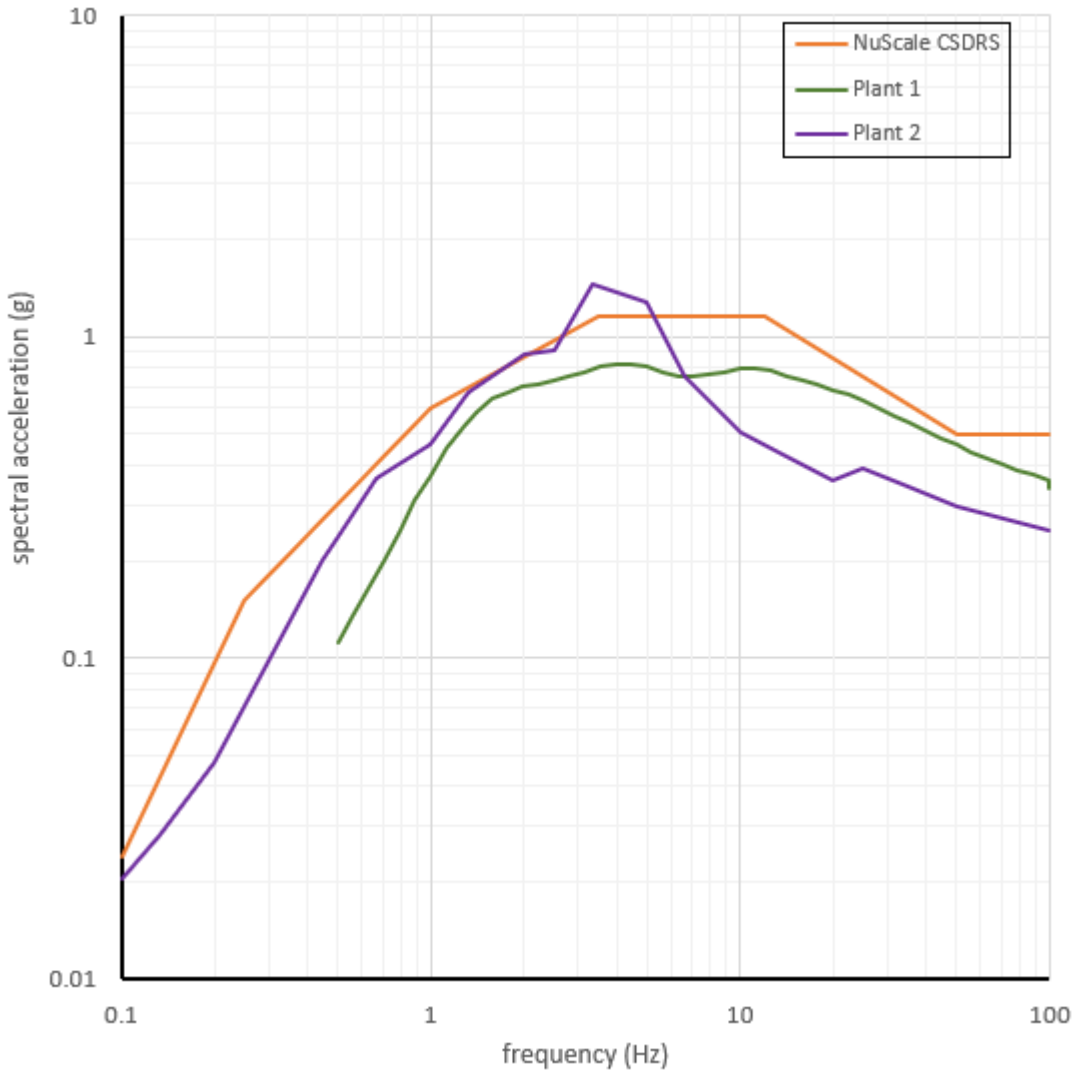
Near-Term Task Force Recommendation 2.1 (R2.1) Seismic Hazard Curves

- For each of the NPP sites licensees performed PSHA to develop seismic hazard curves
  - Seismic Source Characterization Model
  - Seismic Ground Motion Characterization Model
  - Site Response Evaluation based on dynamic properties of local site geology

Estimated Structural SSC Failure Frequency

- Convolve seismic hazard with SSC seismic fragility to determine failure frequency
- Use 0.5, 1.0, and 2.5 Hz seismic hazard curves along with adjusted SSC HCLPF values
- Average SSC failure frequencies for 0.5, 1.0, and 2.5 Hz to determine composite failure frequency

Response Spectra Comparisons



Results for NuScale Structures

Component	HCLPF (g) Table 19.1-35	Assumed Consequence DCA Table 19.1-35	Estimated Seismic Failure Frequency Plant 1	Estimated Seismic Failure Frequency Plant 2
Reactor Building Crane	.88g	Core Damage (CD)/ Large Release (LR)	2.61E-07	6.02E-07
Reactor Building Exterior Walls	.92	CD/LR	3.38E-07	8.91E-07
NPM Supports	.92	CD/LR	3.15E-07	8.26E-07
Pool Walls	.95	CD/LR	2.58E-07	6.72E-07
Crane Support Walls	1.23	CD/LR	1.01E-07	2.56E-07
Bay Walls	1.31	CD/LR	8.63E-08	2.17E-07
Roof	1.20	CD/LR	1.52E-07	3.95E-07
Basemat	1.38	CD/LR	5.33E-08	1.31E-07

Notes

- Each key structure that screens-in would represent a separate seismic sequence.
- Staff is open to alternate ways of handling seismic core damage risk.
- Staff needs to ensure that all credible seismic accident scenarios are being considered.